ACCESSION #: 9211050318
LICENSEE EVENT REPORT (LER)

FACILITY NAME: Oconee Nuclear Station, Unit 3 PAGE: 1 OF 9

DOCKET NUMBER: 05000287

TITLE: Control Rod Groups Drop While Troubleshooting Due To Unknown Cause Results In An Automatic Reactor Trip
EVENT DATE: 09/29/92 LER #: 92-04-00 REPORT DATE: 10/29/92

OTHER FACILITIES INVOLVED: DOCKET NO: 05000

OPERATING MODE: N POWER LEVEL: 73

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
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COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: No

ABSTRACT:

On September 29, 1992, at 0916 hours, Unit 3 tripped from approximately 4% Full Power (FP) following a rapid reduction in power from approximately 73% FP after the Regulating Control Rod Groups dropped into the core. Prior to this transient Instrument and Electrical (I&E) technicians were troubleshooting a problem in the Control Rod Drive (CRD) system. At 0916 hours, I&E technicians opened a CRD breaker associated with the 3A CRD power supply and Control Room Operators observed that a group of Control Rods fell into the core. The Transient Monitor system indicated Groups 5, 6, and 7 dropped into the core. Approximately 13 seconds later the Reactor tripped on low Reactor Coolant System pressure. The unit post-trip response was normal and was stabilized at hot shutdown conditions. The root cause of the event is Unknown: Possible Equipment Malfunction with the 3B CRD power supply. Corrective actions include replacing a programmer, revising procedures and installing a modification.

END OF ABSTRACT

BACKGROUND

The Unit 3 reactor core [EIIS:AC] has 69 control rods [EIIS:ROD] divided into eight groups. Groups 1 through 4 are safety rods and placed in the full out position but when inserted provide sufficient negative reactivity to assure adequate shutdown margin. Groups 5 through 7 are regulating rods and are used to control reactor power. Group 8 rods are the axial power shaping rods and are used to help control the power imbalance in the core within specified limits.

Two separate 575 volt AC power sources and breakers supply two busses with 120 volt AC power. Each bus has a breaker, a transformer to step down the 575 volt AC and a voltage regulator to regulate the 120 volt AC supplied to the Control Rod Drive (CRD) [EIIS:AA] power supplies.

Each bus feeds three power supplies for the CRD system: 1. Regulating 2. DC Hold 3. Auxiliary. Each of the regulating groups (5 through 8) have their own regulating (normal) power supply [EIIS:JX]. It is normally used to hold the rods in place and to move them when a rod motion signal is received. The DC Hold is only required to hold the safety rods (groups 1 through 4) in the full out position, is independent of the regulating and auxiliary power supply and energizes two phases of the stator. The regulating and auxiliary power supplies use solid state programmers to gate (switch) Silicon Control Rectifiers (SCR's), which sequentially energize the six phases of the CRD stator causing rod movement. Each power supply contains its own programmer and rectifiers to produce rod motion. The auxiliary power supply is used to move the safety rod groups (1 through 4) and can also be used to energize the regulating group rods if needed.

The CRD mechanisms use a six-phase, DC stator to raise or lower a lead screw which is coupled to an individual control rod. Two phases of the stator are normally energized to hold a control rod in a fixed position. When the stator is de-energized, the rod will drop on groups 1 through 7.

A position indicator panel is mounted on the vertical board above the diamond control panel in the control room. It has rod "out limit" (red) lamps, "in limit" (green) lamps, "fault" (yellow) lamps, and "on control" (white) lamps.

The Reactor Protective system (RPS) [EIIS:JC] is a safety related system which monitors parameters related to the safe operation of the plant. The RPS provides a two-out-of-four logic for tripping the reactor when a predetermined set point is exceeded. One of the set points is for a low Reactor Coolant System pressure of 1810 psig. This is accomplished via the reactor trip module relays [EIIS:RLY] which de-energize the control rod drive system breakers causing rods to drop.

The Anticipated Transient Without Scram (ATWS) Diverse Scram System (DSS) consists of two channels. The system is normally "Enabled" with input into each channel from the Wide Range pressure transmitters. The channels will produce a trip signal when the associated pressure signal reaches 2450 psi. It takes both channels tripping to produce a CRD "trip" signal. This de-energizes rod groups 5, 6, 7 and the auxiliary power supplies. DSS also sends a signal to raise the turbine bypass valve set point.

EVENT DESCRIPTION

On September 28, 1992, Unit 3 was returning from a refueling outage, at approximately 42 percent Full Power (FP) and holding for the completion of testing. Instrument and Electrical (I&E) personnel began investigation into a problem with the 3A Control Rod Drive (CRD) power supply voltage regulator which was making an unusual chattering noise. During troubleshooting, the I&E personnel placed the voltage regulator in manual and found that the CRD auto transformer voltage regulator drive motor would not move in either direction. I&E technicians opened the CRD breaker number 10 at 0945 hours on September 28, 1992, and replaced the motor. Also, the control circuits had to be replaced due to damage from the motor being seized. The CRD breaker 10 was closed at 1505 hours, however, the motor would only work in manual. Since it was near the end of the shift, and it would not affect the power supplies to have the 3A in manual control and the 3B in automatic control, it was determined the work could be completed on the next day. This was discussed with the operations personnel prior to the end of the shift.

On September 29, 1992, with Unit 3 at approximately 73 percent FP and holding for further testing, a re-clearance to begin work was issued by Unit Supervisor A at 0900 hours.

Although, unrelated to this event I&E took the Honeywell 45000 Operator Aid Computer out of service at approximately 0913 hours, because of monitor problems. This resulted in the Unit 3 Alarm Typer being out of service during this event.

The correct component verification was performed by I&E Technicians A and B at 0913 hours. The technicians had discussed the work to be performed and had all the tools and parts assembled at the CRD breaker cabinet. I&E Technician A went to system Logic Cabinet 2 to verify and balance the voltage between the 10 and 11 CRD breakers. He then went to the Unit 3 CRD system DC Hold Supply Cabinet and measured the current on the breakers. An adjustment was made by I&E Technician B at the Unit 3 CRD Voltage Regulator A Cabinet as Technician A observed the current meter. Technician A then went back to the system Logic Cabinet and rechecked the voltage. At approximately 0915 hours, I&E Technician A went to a phone, approximately 20 feet from the 3A (breaker 10) and 3B (breaker 11) cabinets and informed the Unit 3 Control Room Reactor Operator A (RO-A) that he was ready to open the 3A CRD breaker 10. RO-A concurred and began to observe the stat-alarm panel. At approximately 0916 hours I&E Technician A opened the breaker.

RO-A received and acknowledged the CRD breaker stat-alarm and immediately turned to the rod position indicator panel. He noticed that a rod group had dropped (indicated green on the Position Indicator Panel). He informed the Control Room Senior Reactor Operator (CR-SRO) that a group of rods had dropped. The CR-SRO had observed that group 5 dropped and informed RO-A to manually trip the reactor.

I&E technicians A and B noted that the 3B CRD breaker 11 tripped after only a few seconds from the time I&E Technician A had opened breaker 10.

The reactor automatically tripped at 0916:28 hours, from approximately 4% FP after a rapid power reduction from approximately 73% FP, on low Reactor Coolant system (RCS) pressure. This was just prior to the manual trip performed by RO-A. The 3B CRD breaker number 11 opened automatically, due RPS channels tripping, approximately 13 seconds after the 3A CRD breaker 10 was opened manually. The post-trip scan of the control rod indicators in the control room showed that all rods were at the "in limit" (lights green). The Transient Monitor plots indicated that all regulating rods (groups 5, 6, and 7) initially dropped instead of only group 5.

Post-trip response was as expected. Pressurizer level decreased to 74 inches, then increased to 221 inches. RCS pressure dropped to 1799 psig, then increased and controlled at 2147 psig. An additional High Pressure Injection pump B was manually started and valve 3HP-26 was throttled at 0916:51 hours. The pump was stopped and the valve closed at 0918:57 hours. RCS Hot and Cold Leg temperatures converged and stabilized at approximately 555 degrees F. Steam Generator levels dropped to 22 inches, then stabilized and controlled at the normal low level set point of 25 inches on the startup range. Steam Generator pressures peaked at 1051 psig, then dropped and controlled at approximately 1005 psig.

The Main Feedwater (FDW) [EIIS:SJ] PUMP minimum recirculating valves (3FDW-53 and 3FDW-65) were in the closed position and would not operate. These valves have automatic and manual modes of operation. In automatic, the valves will open automatically to provide a minimum recirculating flow of 2300 gpm to the unit Hotwell [EIIS:KA]. The controller will automatically ramp the flow from 660 gpm to 2300 gpm if the FDW pump hydraulic oil and discharge pressure are above the minimum set points. In manual, the operator positions the valves manually from a controller. Operators took manual control of the valves and an investigation was initiated to determine the cause of the automatic mode failure.

The Unit was stabilized at hot shutdown conditions after the trip. No Engineered Safeguards systems [EIIS:JE] or pressurizer relief valve actuation occurred.

After investigations and the post-trip review were completed the permission to startup was granted on September 29, 1992, at 2015 hours. The reactor was critical at 2357 hours.

CONCLUSIONS

The investigation immediately following the trip on low Reactor Coolant System pressure due to group 5 or group 5, 6, and 7 rods dropping was initiated. Instrument and Electrical (I&E) technicians A and B were familiar with the Control Rod Drive (CRD) system and are qualified to perform work on the system. The day before the trip I&E had worked on the system with CRD breaker 10 open for approximately 8 hours with no adverse consequences.

Transient Monitor System (TMS) data was assembled on previous trips where one group of rods dropped first and where all three regulating rod groups dropped. The TMS clearly differentiates the dropping of one group of rods versus the dropping of three groups. The TMS was also checked for proper operation following this event.

Therefore, it is concluded that all three groups dropped based on the available post-trip data.

For an event where all regulating groups dropped two possible scenarios were analyzed.

A voltage fluctuation occurred that resulted in a low enough signal to lose the gating Silicon Control Rectifiers. The gate drives were checked when powered up and cabinets and components associated with CRD breakers 10 and 11 were inspected for loose connections or crimped wires, however none were found. A temporary modification will be installed to measure and trend voltage input to a rod group. This should identify if the regulating groups are experiencing any voltage fluctuations.

A Diversified Scram System (DSS) spurious actuation would result in rod groups 5, 6, and 7 dropping. A printout of the Anticipated Transient Without Scram (ATWS) events was obtained. The events are recorded and stored in a programmable logic controller register. However, this printout did not reveal if DSS actuated due to a problem in the program. Currently the program stores 99 events and does not allow any event after 99 to be recorded. The program was intended to allow the latest event in and push the oldest event out. However, the stack does not move. Therefore, the registers become full during the testing of ATWS using the I&E procedures and a DSS spurious actuation could not be ruled out. This is possible but not as likely as a voltage fluctuation.

The inspection was also performed in the area concerning group 5 CRD components and wiring. ES-A and I&E Technician A confirmed that the only possible reasons a single group of rods could drop prior to other groups dropping would be the rod group's programmer or the F contactor (electronic trip). These were checked and inspected on each of the three regulating rod groups and no problems were found.

The Feedwater (FDW) Pump Minimum Recirculating Valves (3FDW-53 and 3 FDW-65) failed to open when required. The valves had been changed out and satisfactorily tested during the refueling outage. A controller supplies an air signal to a high gain relay. The relay provides air to a positioner which is designed to position the valve operator as required by the demand for the valve to open or close. The controller is remotely mounted and the relay is attached to the positioner to provide a variable air supply based on the demand. The relay is set by the manufacturer to reset at 5.4 psig. If there is not enough air supply to the relay (at least 5.4 psig) it will not reset resulting in no air supply being sent to the positioner. It was concluded the controller was not supplying an adequate air pressure to reset the high gain relay. The relay was not reset therefore, the positioner did not have an air signal and the valve operator did not operate the valves. The high gain relay has been bypassed by adjusting to an acceptable lower pressure (3 psig) and the investigation is continuing. Operations personnel provided minimum recirculation for the FDW pumps using manual mode of operation.

The root cause of this event is Unknown: possible Equipment Malfunction.

A review of events over the past two years has shown that Unit 3 tripped following a dropped control rod group on November 13, 1990, April 1, 1991, and June 9, 1991.

The November 13, 1990 event occurred due to a programmer failure in the group 7 power supply. The power supply failed during normal operation and group 7 rods dropped into the core. The reactor was manually tripped.

In the event described in this report the programmer was evaluated and found satisfactory.

The rod groups dropped for the April 1, 1991 and June 9, 1991 events while testing was in progress. The April 1, 1991 event involved a Design Deficiency, defective circuit board on a DSS channel and a Deficient Procedure. The June 9, 1991 event involved equipment malfunction where a failed transfer switch allowed the regulating and auxiliary CRD power supplies to simultaneous energize the group 5 CRD mechanisms.

In the event described in this report the DSS was working properly except for the retention of the stored data. The design of the CRD system allows for one CRD AC breaker to be open while the other AC breaker supplies power to the CRD system. Also, this event did not involve the transfer switches associated with the regulating and auxiliary power supplies.

Therefore, based on a review of these three events, this event will be classified as not recurring.

No personnel were injured in this event and there was no release of radioactive materials or personnel overexposures involved.

CORRECTIVE ACTIONS

Immediate:

1. Operations personnel stabilized the unit at hot shutdown and took manual control of the Main Feedwater Pump minimum recirculating valves.

Subsequent:

1. Revise Instrument and Electrical procedures to require verification that voltage is getting to the gate drives down stream of the E and F contactors. Require the use of the procedure when tripping Control Rod Drive breakers that could result in a reactor trip.

2. The programmer associated with group 5 control rods was replaced.

3. Feedwater Pumps Recirculating Valves (3FDW-53 and 3FDW-65) high gain relays were bypassed.

Planned:

1. Install a temporary modification to measure and trend the voltage input to a rod group.

2. Revise the Instrument and Electrical test procedure to ensure the Diversified Scram System programmer is reset when testing is complete.

3. Identify and resolve the air supply problem to the FDW pumps recirculating valves high gain relays.

SAFETY ANALYSIS

Unit 3 tripped from approximately 4% Full Power (FP) on September 29, 1992, following an immediate reduction in power from approximately 73% FP. This happened as a result of regulating rod groups dropping into the core. The Unit automatically tripped on low Reactor Coolant system (RCS) pressure (approximately 1810 psig) approximately 13 seconds after the rods dropped. The plant response to this event was normal and as expected. No Engineered Safeguards system of emergency feedwater actuations were either required or received.

Only the dropping of one control rod is analyzed in the Final Safety Analysis Report, section 15.7, "Control Rod Misalignment Accidents". It is felt that the dropping of a group of rods, while not analyzed, would make it very difficult for the reactor to successfully run back to a lower power level and not trip. The manual or automatic trip of the reactor terminates the initial transient and prevents the reactor from exceeding monitored parameters. Station Operating procedures require the manual trip of the reactor if more than one control rod drops. Operators were in the process

of manually tripping the Unit on september 29, 1992 when the Reactor Protective System automatic trip was initiated. Reactor tilt/imbalance related problems (caused by a group drop) are less significant than the consequences of a single rod drop. This is due to the distribution of the group rods in the core.

There were no personnel injuries, no releases of radioactive materials, or overexposures associated with this event. The health and safety of the public was not jeopardized due to this incident.

ATTACHMENT 1 TO 9211050318 PAGE 1 OF 1

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DUKE POWER

October 29, 1992

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

Subject: Oconee Nuclear Site
Docket Nos. 50-269, -270, -287
LER 287/92-04

Gentlemen:

Pursuant to 10 CFR 50.73 Sections (a)(1) and (d), attached is Licensee Event Report (LER) 287/92-04, concerning control rods dropping, which resulted in an automatic reactor trip.

This report is being submitted in accordance with 10 CFR 50.73 (a)(2)(iv). This event is considered to be of no significance with respect to the health and safety of the public.

Very truly yours,

J. W. Hampton

Vice President

/ftr

Attachment

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